NON-PUBLIC?: N
ACCESSION #: 9108260275
LICENSEE EVENT REPORT (LER)

FACILITY NAME: THREE MILE ISLAND, UNIT 1 PAGE: 1 OF 04

DOCKET NUMBER: 05000289

TITLE: LOW RCS PRESSURE REACTOR TRIP DUE TO CONTROL ROD GROUP 7 DROP
EVENT DATE: 07/24/91 LER #: 91-002 REPORT DATE: 08/23/91

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 92

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: W.G. HEYSEK, TMI-1 LICENSING TELEPHONE: (717) 948-8191
ENGINEER

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: AA COMPONENT: JX MANUFACTURER: D150
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On July 24, 1991 at 0520 the reactor tripped on low RCS pressure due to control rod Group 7 dropping into the reactor core as a result of personnel error in combination with an equipment failure. During the performance of a weekly Reactor Protection System surveillance procedure, an Instrument and Controls technician failed to follow the procedure which resulted in a failure to re-energize the primary power supply to the rod drive programmer assemblies. Failure of the "A" phase gate drive of the Group 7 control rods on the secondary power supply, resulted in a degradation of that power supply. During restoration of the Integrated Control System to AUTO upon completion of the surveillance, the Group 7 control rods received a command to move. The primary power supply remained in a de-energized state and the programmer cycled through the failed gate drive on the secondary group power supply. Due to the inadequate number of phases on the secondary group power supply to

support rod movement, the Group 7 control rods dropped into the core. The Reactor Protection System inserted the remaining rod groups when RCS pressure decreased to 1900 psig.

The event was reported per 10 CFR 50.73(a)(2)(iv).

END OF ABSTRACT

LOW RCS PRESSURE REACTOR TRIP DUE TO CONTROL ROD GROUP 7 DROP

I. PLANT OPERATING CONDITIONS BEFORE THE EVENT

TMI-1 was operating at approximately 92% power. Reactor power was limited by level in the "B" steam generator (OTSG). Instrument and Controls (I&C) technicians were performing weekly surveillance procedure (SP) 1303-4.1 on the "C" channel of the Reactor Protection System. The Integrated Control System (ICS) was operating in full automatic until it was required to be placed in manual per the surveillance procedure.

II. STATUS OF STRUCTURES, COMPONENTS OR SYSTEMS THAT WERE INOPERABLE AT
THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT.

The Reactor Protection System (RPS) weekly surveillance was in progress on the "C" RPS channel in accordance with SP 1303-4.1. The following ICS stations were in manual control to support the RPS testing: Diamond, Reactor Master, Feed Water Loop A/B Masters and Steam Generator/Reactor Master. It was not known prior to the event, however, that the secondary side gate drive circuit for control rod Group 7 power supply phase "A" was not functional.

III. EVENT DESCRIPTION

At approximately 0517, an I&C technician reported to the operating crew that Control Rod Drive (CRD) trip breaker testing AA/BRK!, in accordance with SP 1303-4.1, was complete and the ICS could be returned to full automatic. The operating crew verified that there were no existing abnormal indications on the "C" RPS channel. At 0520, the Diamond and Reactor Master stations JD/PL! were returned to automatic operation. Within 8 seconds of the return to automatic operation, control rod Group 7 dropped into the core. This caused a decrease in reactor power to approximately 40%.

The reactor tripped on low Reactor Coolant System (RCS) pressure sixteen seconds after the rod group dropped. Post trip response was considered normal. As part of the follow-up actions for the reactor trip procedure, the Control Room Operators (CROs) were required to lower OTSG SG/AB! pressure in order to insure that the main steam safety valves SB/PCV! were completely seated. Also, as part of the follow-up actions, the CROs tripped FW-P-1B SJ/P! because the control valves SB/SCV! for the feed pump turbine stuck and would not respond to a signal to reduce speed. Neither of these conditions had any adverse affect on the plant post trip response.

The reactor trip, on low RCS pressure, occurred as a result of the CRD Group 7 control rods dropping into the core. The rod drop was due to a combination of personnel error and equipment failure. The I&C technician failed to properly execute the surveillance procedure in that he marked a step which was required to be performed with an "N/A". In addition, the control room supervisor and

crew failed to perform normal verification of redundant indications which led to the failure to recognize abnormal indications on the Diamond panel and the computer CRT display. The primary power supply AA/JX! had been de-energized as required by the surveillance procedure, but mistakenly had not been re-energized. In addition, the "A" phase gate drive on the secondary power supply was not functional due to a previously unknown failure. The CRD System is designed to operate on only one of the power supplies. The CRO restored the ICS to automatic by placing the Diamond control station in AUTO. When the Group 7 control rods received a command to move, the primary power supply was not energized, and the group programmer cycled through the failed "A" phase gate drive on the secondary power supply. Due to the inadequate number of phases on the secondary power supply to support rod movement, the Group 7 rods dropped into the core.

IV. COMPONENT FAILURE DATA

The nature of the failure in the CRD Group 7 power supply was such that it was not readily apparent and, therefore, could have gone undetected indefinitely while the unit was operating. The "A" phase gating circuit was not functional due to loose connections between two individual transistors and their socket mounts. The failed equipment in the CRD Group 7 secondary power supply was repaired and all of the gating circuits in CRD Group 7 and the remaining group power supplies were verified functional.

V. AUTOMATIC OR MANUAL INITIATED SAFETY SYSTEM RESPONSES

All safety systems functioned in accordance with their design. Upon detection of the low RCS pressure signal, the RPS cabinets tripped causing control and safety rod group insertion.

VI. ASSESSMENT OF THE SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

All safety systems performed as designed. There were no adverse safety consequences as a result of this incident. The CRD Group 7 rod drop resulted in a low pressure reactor trip with a normal post-trip response.

VII. PREVIOUS EVENTS OF A SIMILAR NATURE

None.

VIII. CORRECTIVE ACTION PLANNED

The affected components in the CRD system were repaired as identified in section IV. All maintenance personnel were trained on the use of "N/As" in surveillance and other procedures. The RPS surveillance procedure was revised to clarify the intent of the wording of the specific proceduralized step involved. Additionally, the biweekly CRD surveillance procedure was revised to

require a verification that each gate drive unit fires properly while the control rods are being moved. The Operations Director counselled the CRO and SRO involved in the event individually to ensure their proper understanding of the importance of verifying plant conditions prior to changing equipment status during critical evolutions. Maintenance supervision counselled the responsible I&C technician on the importance of proper procedure understanding and use.

* The Energy Industr
Identification System (EIIS), System Identification
(SI) and Component Function Identification (CFI) Codes are included in brackets, "SI/CFI!", where applicable, as required by 10 CFR 50.73(b)(2)(ii)(F).

ATTACHMENT 1 TO 9108260275 PAGE 1 OF 1

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U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: Three Mile Island Nuclear Station, Unit I (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
LER 91-002-00

This letter transmits Licensee Event Report (LER) No. 91-002-00 which addresses the July 24, 1991 low RCS pressure reactor trip resulting from a control rod Group 7 rod drop. Public health and safety were not affected.

This LER is being submitted pursuant to 10 CFR 50.73. The attachments include NRC Form 366 which provides a brief description of the event while a complete event description is reported on the Form 366A.

Sincerely,

T. G. Broughton
Vice President & Director, TMI-1

WGH:

Attachment

cc: Administrator, Region I
TMI-1 Senior Project Manager
TMI-1 Senior Resident Inspector

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